NON-PUBLIC?: N
ACCESSION #: 9401120226
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Fort Calhoun Station Unit No. 1 PAGE: 1 OF 6

DOCKET NUMBER: 05000285

TITLE: Reactor Trip Due to Turbine Trip on Low Hydraulic Fluid
Pressure
EVENT DATE: 12/06/93 LER #: 93-018-0 REPORT DATE: 01/05/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Craig E. Booth, Shift Technical Advisor TELEPHONE: (402) 533-6874

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: ISV MANUFACTURER: C684
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 6, 1993 at approximately 0227 hours, the main turbine and the reactor tripped during Electro-Hydraulic Control (EHC) pump start testing.

The main turbine trip and resultant reactor trip occurred because of the configuration of the EHC System hydraulic fluid tubing connected to pressure switches PS-5096, PS-5097, and PS-5098, and to associated solenoid valves PS-5097-20 and PS-5098-20. The tubing configuration had been changed during the recent refueling outage by Facility Change Engineering Change Notice (ECN) 93-162. The Root Cause of this event was determined to be inadequate design of the ECN. Specifically, personnel involved in the design and review of the ECN failed to analyze the effects of increased lengths of hydraulic tubing on the operation of the EHC system. The contributing cause is inadequate pre-operational testing

specified in the ECN.

Corrective actions include correcting the configuration of the EHC System tubing; establishment of guidance to identify which Facility Change ECNs have the potential to impact plant safety or reliability, and to specify the appropriate level of review by plant personnel for these ECNs; procedural changes to ensure proper pre-operational testing; and training for appropriate personnel.

END OF ABSTRACT

BACKGROUND

Fort Calhoun Station (FCS) is a two-loop CE design pressurized water reactor rated at 1500 MWt. The Turbine/Generator system was supplied by General Electric. This system includes the turbine Electro-Hydraulic Control (EHC) system.

The purpose of the EHC system is to control the speed and load of the main Turbine-Generator unit by positioning the turbine Stop Valves, Control Valves, and Combined Intermediate Valves based on various feedback parameters. The EHC system consists of the Hydraulic Power Unit (HPU), the Turbine Control Panel (CB-10,11), the hydraulic valve actuators or Control Pacs, the Speed Control, Load Control, and Flow Control Circuits (located in AI-50), and various feedback and actuation devices located on the Turbine and related equipment.

The HPU consists of a hydraulic fluid reservoir, two constant pressure hydraulic pumps (EHC-3A & EHC-3B), six hydraulic accumulators, two fluid coolers, two pump test solenoid valves (PS-5097-20 & PS-5098-20), two automatic pump start pressure switches (PS-5097 & PS-5098), two pump start annunciator pressure switches (PS-5099 and PS-5100), one low hydraulic pressure trip switch (PS-5096), one low hydraulic pressure annunciator switch (PS-3324), and va
ious valves, piping, tubing, and
associated devices. EHC-3A and EHC-3B are connected to a common run of 1.5-inch diameter pipe that serves as the main EHC hydraulic header. On each end of the header is a manifold block, and three hydraulic accumulators are connected to each manifold block. The tubing that supplies hydraulic pressure to the valve Control Pacs is also connected to the manifold blocks.

Certain non-safety related plant design changes and most substitute replacements at Fort Calhoun Station are accomplished by the Engineering

Change Notice (ECN) process. This process is administratively controlled by Production Engineering Division Quality Procedure PED-QP-2, "Configuration Change Control."

In March 1993, ECN 93-162 was initiated to replace four pressure switches (PS-3324, PS-5096, PS-5097, and PS-5098) on the EHC system with Barksdale model B2T-M32SS pressure switches. The change was needed because the design deadband of the original pressure switches was greater than the allowable setpoint/reset point span for the devices.

When the new pressure switches were installed during an outage in May 1993, operability testing revealed that the new pressure switches could not operate properly due to the high vibration transmitted from the EHC pumps to the EHC Hydraulic Power Unit. At that time the old (Model B2T-F32SS) pressure switches were reinstalled in the system until a solution could be determined.

In the original configuration, the tubing supplying the pump start pressure switches (PS-5097 and PS-5098) and the Low Hydraulic Pressure Trip Switch (PS-5096) was connected to a common 0.25-inch fitting in the HPU which was in turn supplied from the main EHC hydraulic header via an eight feet long run of 0.25 inch diameter tubing. A revision to ECN 93-162 was subsequently designed and approved to remove all six of the pressure switches (PS-3324, PS-5096, PS-5097, PS-5098, PS-5099, and PS-5100) from the HPU and mount them on the oil reservoir room wall, located near the HPU. This change, as implemented during the Fall 1993 refueling outage, increased the lengths of the tubing runs between the common fitting and the pressure switches from approximately three feet to approximately 22 feet.

EVENT DESCRIPTION

On December 6, 1993 at approximately 0227 hours, a non-licensed operator was performing procedure OI-ST-10, Attachment 6, Section 1.0 (EHC pump start testing). This was the first time this routine procedure section had been performed following the Fall 1993 refueling outage. The operator had pressed the test pushbutton for pump EHC-3A, which started. Normal conditions were observed locally and in the control room. Approximately five to fifteen seconds after the pump started, the turbine and the reactor tripped. The plant trip occurred approximately four to five seconds after the EHC pump test pushbutton was released. Both EHC-3A and EHC-3B were still running at the time of the trip. The Emergency Response Facilities computer reported that the turbine had tripped on LOW HYDRAULIC FLUID PRESSURE.

At 0233, Standard Post Trip actions of EOP-00 and event diagnostics were completed. The Senior Reactor Operator directed entry into Reactor Trip Recovery procedure, EOP-01. At 0239 an equipment operator reported to the control room that suction valve FW-318 for feedwater pump FW-4B was leaking from what was later determined to be the body-to-bonnet joint. As FW-4B was the only main feedwater pump in operation following the actions of EOP-00, the control room operators initiated auxiliary feedwater to the Steam Generators using the motor driven auxiliary feed pump FW-6. With auxiliary feedwater established, FW-4B was secured. The emergency diesel generators were then secured, 345Kv backfeed to non-vital 4160v buses 1A1 and 1A2 was established, and shutdown margin was verified.

At 0439, with plant conditions stabilized and the exit conditions of EOP-01 met, a transition was made to procedure OP-1, Master Checklist for Plant Startup. Troubleshooting the cause of the trip switch (PS-5096) actuation was initiated.

It was theorized that a pressure dip resulted when the test pushbutton for EHC-3A was released. Testing after the plant trip confirmed that pressure dips occurred when using the pump test pushbutton for auto-start of the EHC pump.

It was further theorized that the pressure dip was affecting the Low Hydraulic Pressure Trip Switch (PS-5096) because the tubing that supplied pressure to PS-5096 was connected to a common 0.25 inch fitting with the tubing that supplied pressure to PS-5097 and PS-5098. This fitting was supplied from the EHC system header via a 0.25 inch diameter run of tubing, which could act as a restriction and allow hydraulic pressure in the tubing to momentarily drop below hydraulic pressure in the EHC system header.

This theory was tested by disconnecting the tubing that supplies PS-5096 from the common 0.25 inch fitting and reconnecting it, via a temporary tubing jumper, to the main EHC System header. The EHC pump auto-start testing was then repeated. The test was performed three times and each time no pressure dip was recorded and no actuation of the Low Hydraulic Pressure Trip Switch occurred.

Based on the troubleshooting noted above, it was concluded that the act of performing EHC pump start testing caused a pressure transient that actuated PS-5096, resulting in a turbine/reactor trip. This was due to the extra length of the instrument tubing (added by ECN 93-162) which supplied the EHC Pump Auto-Start Pressure Switches (PS-5097 and PS-5098) and the fact that this tubing was interconnected with the Low EHC Pressure Trip Switch (PS-5096).

The NRC was notified of the event on December 6, 1993 at 0329 pursuant to 10 CFR 50.72(b)(2)(ii). This LER is submitted pursuant to 10 CFR 5073(a)(2)(iv).

EVALUATION/SAFETY ASSESSMENT

The event was significant in that the turbine trip and resultant reactor trip presented an unnecessary challenge to the plant systems, and resulted in an unplanned plant outage.

The RPS was the only safety system that was actuated when it provided a Loss of Load trip signal. Other than the Emergency Diesel Generators starting and running at idle as expected, no ESF system actuated during the event and no primary or secondary safety relief valves lifted. All systems responded as designed. The operator transferred from main feedwater to auxiliary feedwater due to the leaking suction valve FW-318. The event did not pose a danger to the health or safety of the public.

CONCLUSION

The main turbine trip and resultant reactor trip occurred because of the configuration of the EHC System hydraulic fluid tubing connected to pressure switches PS-5096, PS-5097, and PS-5098, and to solenoid valves PS-5097-20 and PS-5098-20. The tubing configuration had been changed through implementation of ECN 93-162 during the Fall 1993 refueling outage.

The root cause of this event was determined to be inadequate design of ECN 93-162. Specifically, personnel involved in the design and review of ECN 93-162 failed to analyze the effects of the increased lengths of hydraulic tubing on the operation of the EHC system.

A contributing cause of this event was inadequate pre-operational testing specified in ECN 93-162. The leak checks and calibration procedures specified did not properly demonstrate operability of the system. This involves both human performance and procedural issues, as PED-QP-2 directs Design Engineering personnel to specify testing requirements, but does not clearly refer to the specific testing requirements in Standing Order G-21, "Modification Control."

The probable cause of the leakage from valve FW-318 was determined to be a pressure surge in the feedwater train when the turbine/reactor tripped.

CORRECTIVE ACTIONS

The following corrective actions have been or will be completed:

(1) The instrument tubing supplying hydraulic pressure to PS-5096 was configured to adequately separate it from all other instrument tubing and to connect directly to a main EHC hydraulic header manifold.

(2) Appropriate Production Engineering Division personnel have been briefed on this event, with emphasis on adherence to procedural guidance and on attention to detail in design and pre-operational testing development and review.

(3) PED-QP-2 will be revised by January 31, 1994 to clearly state that pre-operational testing requirements and acceptance criteria, in accordance with SO G-21 and any other applicable procedure, must be addressed in Facility Change ECNs. Specifically, potential changes to electro-hydraulic system response, as occurred in this event, will be identified as a criterion to consider during ECN design and testing.

(4) OPPD will establish guidance to identify which Facility Change ECNs have the potential to impact plant safety or reliability, and to specify the appropriate level of review by plant personnel for these ECNs. Appropriate procedural changes to implement this guidance will be completed by February 28, 1994.

(5) Appropriate OPPD personnel will be trained by March 31, 1994 on this event, including corrective actions.

PREVIOUS SIMILAR EVENTS

LER 93-012 discussed inadequately sized cables for motor operated valves which was attributed to lack of attention to detail in the design package preparation. LER 92-022 was referenced as a similar event in LER 93-012.

OPPD
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444 South 16th Street Mall

Omaha, Nebraska 68102-2247
402/636-2000

January 5, 1994
LIC-93-0331

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docke
No. 50-285

Gentlemen:

SUBJECT: Licensee Event Report 93-018 for the Fort Calhoun Station

Please find attached Licensee Event Report 93-018 dated January 5, 1994.
This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv).

Please contact me if you have any questions.

Sincerely,

W. G. Gates
Vice President

WGG/tcm

Attachment

c: LeBoeuf, Lamb, Leiby & MacRae
J. L. Milhoan, NRC Regional Administrator, Region IV
R. P. Mullikin, NRC Senior Resident Inspector
S. D. Bloom, NRC Project Manager

45-5124

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*** END OF DOCUMENT ***